Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/767
AUSTRIA
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Experiments on the Reprocessing of Uranium Fission Product Solutions by the Exclusive Use of Ion Exchangers.

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In the reprocessing of nuclear fuel material, ion exchangers have so far been employed for the seperation of certain elements only. The costliness of exchanger material as well as its instability in the case of high radiation doses have rendered its general use impracticable.

The development of inexpensive cation exchangers on wood basis was made possible by a Research Contract from the IAEA. These exchangers demonstrated good seperation properties for certain fission product elements, and it therefore seemed feasible to examine a complete reprocessing procedure using only ion exchangers.

The maximum loading factor and elution velocities possible for uranium and the main group of fission products were determined in several column experiments.

In the first exchange step, a pre-seperation of uranium from the fission products (Cs^{+}, Sr^{2+}, Y^{3+}, etc.) was carried out using an anion exchanger (Dowex 1 x 8), 4.2 N Aluminium nitrate solution gave good results as elution medium. However, not all elements may be separated from uranium by this first step. So part of cerium-III remains adsorbed with the uranium on the exchanger. In this case, the maximum elution velocities are relatively low.

In the second exchange step, the final separation of uranium from the fission products, which remain adsorbed with it on the anion exchanger, is carried out with a sulfitated cation exchanger on wood basis. The uranium-cerium-III separation can be successfully

obtained with a diluted NH_4NO_3-HNO_3 solution, and high elution velocities are possible, although both elements have similar nitrate complex properties.

In the third exchange step, the fission products are separated singly, also with a cation exchanger on wood basis. Good results were obtained by elution with nitric acid, the concentration of which was stepwise increased. Cs^+, Sr^{2+}, and Y^{3+}, which still contained $Al(NO_3)_3$ from the first exchange step, were separated by elution with 0.015, 0.14 and 0.47 M HNO_3. In this case also high elution velocities can be applied.

The beta and gamma doses were calculated for the exchanger which is irridiated to the greatest extent. For this calculation it was assumed that the daily fuel consumption of a 100 MW_t nuclear reactor has to be reprocessed by means of ion exchangers. The result of the calculation indicates that the process can be easily conducted in such a manner that no significant radiation damage occurs to the exchanger material.

The possibilities of reprocessing nuclear fuel by ion exchangers in small plants can be of advatage to many reactor staions. In addition to this, a complete separation of the fission products lessens the storage problem and gives better possibilities for the technical application of certain fission elements.